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SOUTHERN COMPANY
Energy to Serve Your World℠

NL-05-1264

August 31, 2005

Docket Nos.: 50-348 50-424
50-364 50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Ladies and Gentlemen:

Pursuant to the requirements of Nuclear Regulatory Commission (NRC) Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," issued to the Southern Nuclear Operating Company (SNC) on September 13, 2004, SNC hereby submits Enclosures 1 and 2 which constitute the required September 1, 2005 responses for Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2 and Vogtle Electric Generating Plant (VEGP) Units 1 and 2. Enclosures 3 and 4 contain a list of regulatory commitments for FNP and VEGP respectively.

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

Sworn to and subscribed before me this 31 *day of* August *, 2005.*

Notary Public

My commission expires: ________

NOTARY PUBLIC STATE OF ALABAMA AT LARGE
MY COMMISSION EXPIRES: June 10, 2008
BONDED THRU NOTARY PUBLIC UNDERWRITERS

LMS/chm/sdl

Enclosures:
1. Farley September 2005 Response to Generic Letter 2004-02
2. Vogtle September 2005 Response to Generic Letter 2004-02
3. Farley List of Regulatory Commitments
4. Vogtle List of Regulatory Commitments

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. L. M. Stinson, Vice President – Plant Farley
Mr. D. E. Grissette, Vice President – Plant Vogtle
Mr. J. R. Johnson, General Manager – Plant Farley
Mr. T. E. Tynan, General Manager – Plant Vogtle
RType: CFA04.054; CVC7000; LC# 14306

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. C. A. Patterson, Senior Resident Inspector – Farley
Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 1

Farley September 2005 Response to Generic Letter 2004-02

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 1

Farley September 2005 Response to Generic Letter 2004-02

General System Description:

Farley Nuclear Plant (FNP) Units 1 and 2 are Westinghouse three loop Pressurized Water Reactor (PWR) design. The Residual Heat Removal System (RHR) (low head safety injection), Centrifugal Charging System (CVCS) (high head safety injection) and Containment Spray System (CSS) pumps are started following a Loss of Coolant Accident (LOCA). Initially, two RHR, two CVCS and two CCS pumps take suction from the Refueling Water Storage Tank (RWST). When the RWST level reaches the low level set point, the RHR pumps are manually stopped and are realigned to take suction from the post LOCA containment sump. Once the RHR switchover to recirculation is complete, the CVCS pumps take suction from the RHR pump discharge.

When the RWST level reaches low-low level, the CSS pumps are realigned to take suction from the containment sump. There are four independent suctions (two for RHR and two for CSS) located on elevation 105'-6" in the containment, the lowest floor elevation in the containment exclusive of the reactor cavity, and they are located outside the secondary shield wall.

Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004, requested under "Requested Information," item 2, "Addressees are requested to provide the following information no later than September 1, 2005." The requested information is hereby provided.

GL Item 1:

The 90 day response was provided in SNC letter dated February 25, 2005.

GL Item 2 (a):

Confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.

SNC Response:

Analysis and evaluations are underway to confirm that the Emergency Core Cooling System (ECCS) and CSS recirculation sumps will function under debris loading conditions as specified by Generic Letter 2004-02. FNP will be in compliance by December 31, 2007 with applicable regulatory requirements using methodology identified in Nuclear Energy Institute (NEI) document NEI 04-07, Rev 0, December 2004, "Pressurized Water Reactor Sump Performance Evaluation Methodology" as modified by NRC safety evaluation report issued December 6, 2004.

Major activities planned or completed for SNC include:

Walkdown of containment, for final design change input and to confirm the latent debris figures previously collected, are scheduled and will be completed prior to final screen construction.

Debris generation analysis has been completed. Insulation inside containment that is affected during a LOCA event is mostly Reflective Medal Insulation (RMI) with very little fiber.

Debris transport analysis has been completed showing the debris loading at the sump screens. A Computational Fluid Dynamics (CFD) model was used per the NEI guidance.

Net Positive Suction Head (NPSH) calculations have been completed for the RHR and CCS pumps.

Preliminary screen sizing requirements have been determined per the NEI guidance. Verification testing is scheduled.

Downstream effects evaluation is in progress.

SNC is planning to use a passive engineered screen. Screen installations for both units 1 and 2 are planned in 2007.

Modifications are being considered for the safety injection branch line throttle valves, and the RHR pumps backup seals.

GL Item 2 (b):

A general description of and implementation schedule for all corrective actions, including any plant modifications that you identified while responding to this generic letter. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If all corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.

SNC Response:

SNC has outages scheduled for April 7, 2007 on Unit 2 and September 29, 2007 for Unit 1. Required modifications will be completed during these 2007 outages, and as such all actions to comply with GL 2004-02 will be completed by December 31, 2007.

SNC has an outage for Unit 1 scheduled for April 8, 2006. The start date of this outage was recently moved from April 1, 2006 to April 8, 2006 to support Southern Company System Grid stabilization requirements. The timing of this Unit 1 outage does not allow for completion of the analysis, testing and manufacturing of the required sump screens. Therefore, SNC will not implement the identified actions during the first refueling outage starting after April 1, 2006. This has been discussed with the NRC staff. SNC has scheduled the implementation of required changes during outages for each unit in 2007.

Initial analysis activities for debris generation and transport have been completed following the NEI / NRC Guidance. The results of the analysis have been used to develop initial sizing of engineered passive screens for each sump suction. Additional evaluations are under way by Westinghouse to address the possible downstream effects of debris passing through the sump screens. Preliminary chemical effects analysis has been completed.

The results of the downstream effects may result in modifications to the safety injection (SI) branch line throttle valves. Modifications may also be required to the RHR pumps backup seals. Any plant modifications required as a result of the downstream effects evaluations will be completed prior to December 31, 2007.

Provided below is the schedule of activities for the completion of analysis, testing and installation of the new engineered screens.

Activity	Common	Unit 1	Unit 2
Walkdown of containment for design considerations		Spring 2006	Fall 2005
Debris generation analysis	Complete		
Debris transport analysis (CFD)	Complete		
NPSH calculations	Complete		
Screen sizing per NEI guidance	Complete		
Vendor Testing of screens	Spring 2006		
Downstream effects evaluation including plugging and excessive wear	Spring 2006		
Delivery of engineered screens		Fall 2007	Spring 2007

Activity	Common	Unit 1	Unit 2
Installation of engineered screens		Fall 2007	Spring 2007
Modification to SI Branch Line Throttle Valves, and pumps seals, if required		Fall 2007	Fall 2007
Removal of tags, labels, etc. not qualified for LOCA environmental conditions, as required		Fall 2007	Spring 2007
Revision of procedure for control of signs and labels inside containment	Spring 2007		
Programs for control of materials in containment are in place. (i.e., insulation, coatings and foreign materials)	In place		

GL Item 2 (c):

A description of the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. The submittal may reference a guidance document (e.g., Regulatory Guide 1.82, Rev. 3, industry guidance) or other methodology previously submitted to the NRC. (The submittal may also reference the response to Item 1 of the Requested Information described above. The documents to be submitted or referenced should include the results of any supporting containment walkdown surveillance performed to identify potential debris sources and other pertinent containment characteristics.)

SNC Response:

SNC has performed analysis to determine the susceptibility of the ECCS and CSS recirculation functions for Farley Nuclear Plant to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. These analyses conform to the greatest extent practicable to the NEI 04-07 methodology as approved by the NRC safety evaluation report dated December 6, 2004. Following is a description of the analysis areas performed:

Containment Walkdown

Walkdown of containment was performed by SNC personnel using the guidance of NEI 02-01. The information obtained from the walkdown confirmed the insulation that was installed in containment matched the design documentation. General information was obtained that confirmed the general house keeping condition of containment. For the purpose of the screen sizing analyses, the latent debris was assumed to be 200 lb_m. Additional walkdowns are planned for the next unit 2 outage for latent debris and coatings evaluations. The assumed latent debris and coatings quantities may be adjusted based upon the results of the walkdown.

Pipe Break Characterization

Pipe break characterization was performed by Sargent and Lundy of Chicago. The FNP NSSS system is a three loop Pressurized Water Reactor (PWR). The system consists of one reactor vessel (RPV), three steam generators (SGs), three reactor coolant pumps (RCPs), one pressurizer (PZR) and the Reactor Coolant System (RCS) piping. The NSSS system is located inside a bio-shield and the Reactor cavity. The area inside the bio-shield is mostly open at the lowest levels, with the exception of the reactor cavity and surrounding walls in the center, and a concrete wall between the A and C loops. The concrete wall between loops A and C has a walkway against the reactor cavity wall that allows an opening between loops A and C. The outer bio-shield walls extend from the containment base elevation of 105'-6" to El. 129'-0". There are areas of the bio-shield walls that are partially open; an inner wall extends from El. 105'-6" to 116'-3", and an outer wall extends down from El. 129'-0" to elevation 115'-3" at some locations. Above elevation 129'-0" smaller "vaults" or "coffins" surround each loop and the associated Steam Generator and Reactor Coolant Pump. These "vaults" further narrow around the Steam Generator at El. 155'-0" and extend up to El. 166'-6". There is also a separate "vault" for the Pressurizer that begins at El. 129'-0" and extends up to El. 181'-0".

The piping runs considered for breaks are the RCS hot legs, the RCS cold legs, RCS interim legs, and all RCS attached energized piping. Breaks in these lines could decrease RCS inventory and result in the ECCS and/or CSS operating in recirculation mode, in which the system pumps would take suction from the containment sumps.

The majority of the insulation inside the FNP containment is either Reflective Metal Insulation (RMI) or Transco RMI. The Transco RMI is located on the Steam Generators and small sections of the attached piping. The remainder of insulation is Mirror RMI. There is also a small amount of Tempmat fiber located on the Steam Generator instrumentation reference legs and very small sections of the Reactor Vessel bottom head insulation assembly. In addition, there is a large amount of closed cell foam type (Armaflex) insulation located on the chilled water lines (Service Water and Component Cooling Water). However, the Armaflex insulation has a very low density and will float if dislodged (even if reduced to particles) and not add to the debris mixture on the sump screens which are 100% submerged during recirculation.

The largest energized lines in containment that require evaluation are the hot legs (29-inch ID), the interim leg (31-inch ID), the cold leg (27-1/2-inch ID), the pressurizer surge line (14-inch nominal), RHR recirculation line to the hot leg (12-inch nominal diameter) and safety injection to the cold leg (12-inch nominal diameter). The other piping lines have a much smaller diameter.

Since the RHR recirculation lines and the safety injection lines are located within the bio-shield enclosure and are of smaller diameter than the RCS piping, these line breaks would be bounded by the reactor coolant loop breaks and thus are not analyzed. This leaves breaks in the hot legs, the cold legs, interim legs and the pressurizer surge line for consideration.

The interim leg is the largest line (31-inch ID) within the bio-shield enclosure and would produce the largest zone of influence (ZOI). Placing a break on the interim leg, on the same loop as the pressurizer surge line (loop B), potentially captures the most insulation debris. An interim leg break in loop C is also considered since it will create a large amount of debris and is the bounding location for coating debris. A cold leg break on loop A near RCP discharge is also considered since it can generate a large amount debris from both loops A and B.

A hot leg or cold leg line break at the RPV is also considered. The RPV is covered with mirror RMI insulation and has a small amount of Temp-mat fiber insulation used around the incore instrumentation tubes in the bottom head. This break would affect the reactor insulation and the insulation on the RCS lines adjacent to the break, up to the penetrations. However, this debris would fall to the bottom of the reactor vessel cavity, but would be in a stagnant pool and will not transport to the sump. The amount of debris would also be bounded by a hot or cold line break elsewhere on the line. Therefore, a hot leg or cold leg break at the RPV is not analyzed.

The postulated break locations are as follows:

S1. The Loop C Interim Leg near the base of the steam generator at El. 118'-0" [31-inch ID]
S2. The Loop B Interim Leg near the base of the steam generator at El. 118'-0" [31-inch ID]
S3. The Loop A Cold Leg near the RCP discharge at El. 122'-9" [27.5-inch ID]

Alternate Methodology

For the alternate methodology, the selection of the break size and location in Region I is much simpler. The break size for Region I under the alternate break evaluation is defined as either:

- A complete guillotine break of the largest line connected to the RCS piping (14-inch pressurizer surge line)

OR

- A main loop line break equivalent to a guillotine break of a 14-inch Schedule 160 pipe

After performing several iterations, the S2 break has been found to generate the greatest quantity of debris. For the break S4, according to the methodology, a 14-inch Schedule 160 (11.19 inch ID) double-ended guillotine break is modeled on the Loop B interim leg at the same location as the S2 break.

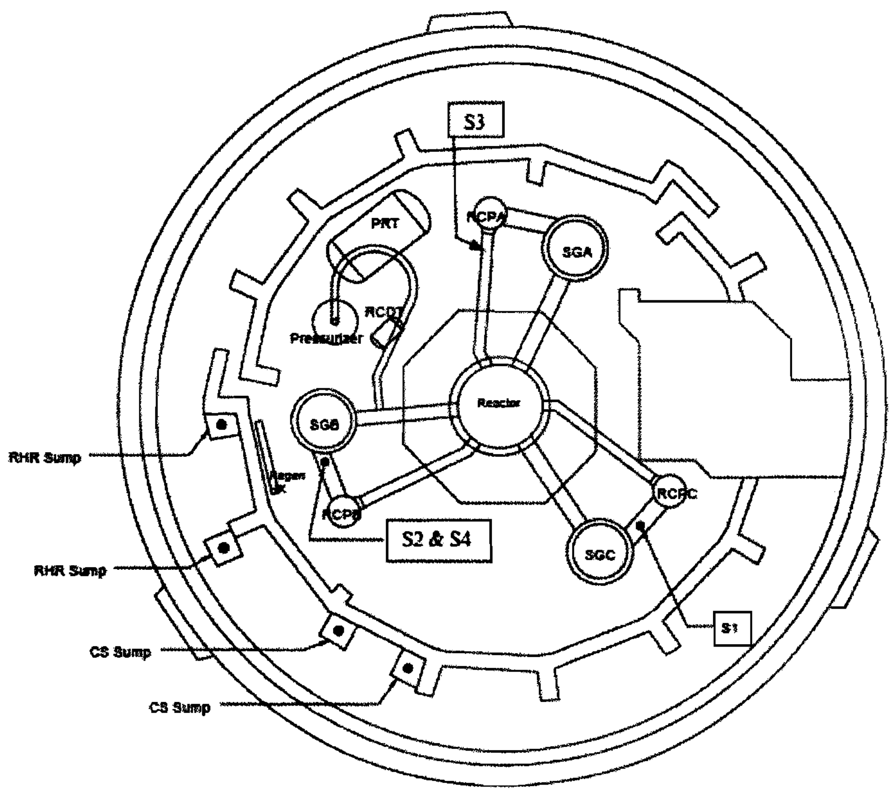
For Region II of the alternate methodology, the debris quantities are the same as for the deterministic methodology. Below is a summary of the postulated break locations.

Break Summary

Postulated Break Locations

Break Name	Break ID	Elevation	Piping
S1	31-inch	118'-0"	Interim Leg – Loop C
S2	31-inch	118'-0"	Interim Leg – Loop B
S3	27.5-inch	122-9"	Cold Leg – Loop A
S4	11.19-inch	118'-0"	Alternate Break (Interim Leg –Loop B)

Postulated Break Locations



Debris Generation

The debris generation analysis was performed by Sargent and Lundy of Chicago. The analysis determined the debris generated based on the NEI guidance and NRC SER of the NEI guidance. The analysis determined the ZOI for each type of material identified inside containment.

Insulation found inside containment that is adversely affected during a LOCA event, was determined to consist of Tempmat fiber, Transco RMI and Mirror insulations. Most of the insulation is Transco RMI and Mirror RMI. The amount of Tempmat fiber is very small.

Qualified coatings are considered in the debris analysis if they are located within a ZOI radius of 10 D.

Unqualified coatings are used on various vendor-supplied components throughout containment. The total square footage of unqualified coatings was determined as part of the NEI 02-01 walkdown. The value determined as a result of this walkdown was relative low. Therefore to apply a conservative value, the option of assuming 10,000 ft^2 was used.

For the purpose of these analyses, the latent debris was assumed to be 200 lb_m. Additional walkdowns are planned for the next unit 2 outage for latent debris verification.

Foreign material (i.e., tags, labels, etc. not qualified for LOCA environmental conditions) may fail following a LOCA and therefore can be transported to the sump. Actions will be taken by SNC to insure that the quantity of foreign material will be minimized. Walkdowns during the next outage on each unit will identify any materials that will require removal during the sump modification outages.

Latent Debris Accumulation within Containment

Programmatic controls are in place at FNP that give bases for the amounts of foreign material and latent debris inside containment remaining below the amounts assumed in the sump analysis. These programmatic controls are described in the response to item 2(f).

Debris Transport to the Sump

A debris transport analysis estimated the fraction of debris that is transported from debris sources (break locations) to the sump screen. The transport analysis is in accordance with the guidance of NEI 04-07 and the applicable NRC SER. The computational fluid dynamics (CFD) analysis was performed by RWDI Consulting Engineers and Scientists for Sargent and Lundy of Chicago. The CFD modeling techniques used are consistent with the SER, NEI Document number 04-07, and NUREG/CR-6773.

CFD analyses of the post-LOCA recirculation flow patterns within the FNP containments were performed to quantify the flow velocities expected inside the secondary shield wall, through the secondary shield wall, outside the secondary shield wall and near the CS and RHR sumps. CFD analysis of the post-LOCA recirculation containment flows following Break S2 indicate velocities of 0.4 to 1.1 ft/sec can be expected in the containment flow field. These velocities are such that debris within the shield wall would be transported to the openings in the wall. Transport velocities through the wall and to the sumps are of the same order of magnitude.

Head Loss as a Result of Debris Accumulation

The engineered sump screens that will be installed at FNP are designed to operate in such a way that the thin bed effect does not occur on the sump screen surface. This is due to the small amount of fiber present in the FNP containment. Parametric analyses were performed to estimate the surface area of the engineered screen that meets the FNP head loss criterion for the identified debris inventory.

Debris Source Term Reduction

Foreign material (i.e., tags, labels, etc. not qualified for LOCA environmental conditions) may fail following a LOCA and therefore can be transported to the sump. Actions will be taken by SNC to insure that the quantity of foreign material will be minimized. Walk downs during the next outages on each unit will identify any materials that will require removal during the sump modification outages.

Sump Structural Analysis

Structural analysis of the engineered passive screen will be completed when the final screen design is completed. A general description of the screen design is given in the SNC response to item 2 (d) (vii).

Upstream Effects of Debris Accumulation

Evaluations of containment along with review of the CFD model indicate no significant areas will become blocked with debris and hold up water during the sump recirculation phase. As a precautionary measure, SNC will inspect the reactor cavity drain covers during the next outage on each unit and determine if modifications are needed.

Downstream Effects associated with any Debris Bypass

The methodologies of NEI 04-07 as modified by the NRC safety evaluation, dated December 6, 2004, and WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," are used to evaluate the downstream effects of debris that is passed by the sump strainer.

GL Item 2 (d):

The submittal should include, at a minimum, the following information:

(i) The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

SNC Response:

Analyses have been completed that show the minimum available NPSH margins with an unblocked sump screen are:

RHR pumps: 3.4 feet
CS pumps: 5.7 feet

These NPSH values include entrance and vortex breaker losses which will be assigned to the engineered screen.

(ii) The submerged area of the sump screen at this time and the percent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation.

SNC Response:

All sump screens will be fully submerged (submergence of 100%) before the switchover to recirculation is initiated. The screens will have a nominal submergence of 3 inches at minimum containment water level.

Based on preliminary analysis results, the submerged area of the sump is approximately:

RHR pumps: 900 square feet
CS pumps: 600 square feet

These areas are based upon very conservative analysis using NUREG/CR 6224 methodology. Planned screen testing is expected to result in significant decrease in screen sizing.

(iii) The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS wash down should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates and chemical precipitants caused by chemical reactions in the pool.

SNC Response:

The intended allowance for head loss from debris accumulation and chemical effects on the submerged sump strainer is 2.0 ft for RHR and 3.0 ft for CS.

Comparisons of the Integrated Chemical Effects Test (ICET) to the FNP plant specific parameters has been performed and show that the ICET Test # 2 parameters are the closest comparison to FNP (Test # 2 used trisodium phosphate (TSP) buffer and fiberglass). Based on evaluation of the results to ICET Test # 2, the FNP design and the FNP debris loading, the screen design was adjusted to include an increase in the size of the screens to accommodate a 10% increase in head loss across the screens due to chemical effects. Sump screen vendors are currently developing test plans to quantify the additional head loss associated with chemical debris. SNC will incorporate the results of the testing into the final design of the screen as appropriate.

The primary constituents of the debris bed (break S2) are provided below.

Debris Type	Units	Quantity Generated
Fibrous Debris		
Fiber Insulation – Tempmat	ft^3	1.0
Total Fiber Debris	ft^3	1.0
Reflective Metal Insulation Debris		
Reflective metal Insulation – Transco	ft^3	2,383
Mirror Insulation	ft^3	35,714
Total RMI Debris	ft^3	38,097
Particulate Debris		
Qualified Coatings	ft^3	42.1
Unqualified Coatings	ft^3	5.0
Total Particulate	ft^3	47.1
Other Items		
Latent Debris (15% fiber and 85% particulate)	lbm	200
Foreign Materials (Labels, Placards, etc)	50 ft^2 sacrificial area*	
Chemical Effects	10 % increase in head loss across the screens	

* may be adjusted based upon results of confirmatory walkdown

(iv) The basis for concluding that the water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flow paths.

SNC Response:

Evaluations of containment along with review of the CFD model indicate no significant areas will become blocked with debris and hold up water during the sump recirculation phase. The area of the refueling cavity, which is the area around the reactor head that is flooded prior to fuel movement, is the only significant area in containment that can retain water during an event that requires containment spray. As a precautionary measure, SNC will inspect the reactor cavity drain covers during the next outage on each unit and determine if modifications are needed. This will further insure that water will drain from the cavity to the recirculation sump.

(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flow paths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

SNC Response:

The methodologies of NEI 04-07 as modified by the NRC safety evaluation and WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," are being used to evaluate the downstream effects of debris that is passed by the sump strainer. These evaluations are being performed for all components in the recirculation flow paths including, but not limited to, throttle valves, flow orifices, spray nozzles, pumps, heat exchangers, and valves. The following components were identified as requiring additional evaluation:

- ECCS and CSS Pumps (Centrifugal Charging Pumps, RHR Pumps, CS Pumps)
- ECCS and CSS valves including the SI Branch Line Throttle Valves
- ECCS orifices and CSS nozzles
- Piping and instrumentation
- Reactor Vessel internals
- Nuclear Fuel Assemblies

These evaluations are expected to be complete by Spring 2006.

The evaluation of the ECCS orifices, CSS nozzles, piping and instrumentation, and reactor vessel internals indicate that flow restrictions will not occur.

The results of the downstream effects may result in modifications to the safety injection (SI) branch line throttle valves. Modifications may also be required to the RHR pumps backup seals. Any plant modifications required as a result of the downstream effects evaluations will be completed prior to December 31, 2007.

Preliminary evaluation of the nuclear fuel assemblies indicate that a thin bed of fiber will not form on the fuel. Adequate core cooling will be maintained.

Adverse gaps or breaches are prohibited by the sump strainer specification, which requires that there shall be no spaces or gaps in the final installation that would allow passage of any particles larger than the screen perforation size.

In addition, technical specifications require that the sump strainers be inspected at least once per 18 months. The inspection procedure currently requires verification that there are no unacceptable holes or gaps in the strainer or between the strainer and adjacent structures and components.

Based on the preliminary reviews performed, the design of the new screens and the associated design modifications being considered, and the surveillance requirements that are in place, inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flow paths downstream of the sump screen.

(vi) Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.

SNC Response:

See response 2(d)(v).

(vii) Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under predicted flow conditions.

SNC Response:

SNC will install an engineered passive strainer on each RHR and CSS containment sump inlet pipe. The screens will be located outside the secondary shield wall between the shield wall and the containment wall and as such are not exposed to jet impingement or postulated missiles generated from a LOCA event. The screens are of a robust design that will support structural and hydraulic load created by the accumulation of debris during the post LOCA environment. This robust design provides the strength of trash racks and is adequate to protect the screen during a LOCA event.

(viii) If an active approach (e.g., back flushing, powered screens) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

SNC Response:

FNP is not using the active approach. This section is not applicable to FNP.

GL Item 2 (e):

A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.

SNC Response:

No additional licensing bases changes have been identified.

GL Item 2 (f):

A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to the extent that their responses address these specific foreign material control issues.

SNC Response:

Insulations

As part of the design change process at SNC, the FSAR, calculations and design specification are reviewed for impacts. Design specifications for thermal insulation for piping and equipment contain the requirements for insulation used inside containment. Design calculations for debris generation will be maintained in the calculations index and thus will be reviewed as design changes are prepared for equipment in containment. The containment sump analysis and a description of the insulation types and amounts assumed in the analysis will be included in the FSAR. Plant procedures control the installation and verification of insulation materials. These procedures provide reference to the guidance provided in the design specifications. Therefore the design change process for SNC will insure that the assumptions for insulation type and quantity inside containment will be maintained within the analysis assumptions.

Signs and Labels

Procedures identify label and sign materials that have been evaluated for use in containment. Procedures prohibit the mounting of signs in containment without a written evaluation. The preparation of an evaluation requires the review of the FSAR that leads to a discussion of debris generation during a LOCA and the consideration of signs and labels in containment. Procedure enhancements will be made to clearly identify labels that label and signs that have been evaluated for use in containment during the LOCA environment.

Coatings

As presented in the SNC response, dated October 23, 1998, to Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment*, controls for the procurement, application, and maintenance of Service Level 1 protective coatings used inside the FNP containments have been implemented in a manner that is consistent with the licensing basis and regulatory requirements applicable to FNP. The requirements of 10 CFR Part 50, Appendix B are implemented through specification of appropriate technical and quality requirements for the Service Level 1 coatings program that includes ongoing maintenance activities.

SNC periodically conducts condition assessments of Service Level 1 coatings inside containment. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement, as necessary. The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings that may be susceptible to detachment from the substrate during a LOCA event is minimized.

Foreign Materials

SNC procedure, "Foreign Material Exclusion Program," establishes the administrative controls and personnel responsibilities for the Foreign Material Exclusion (FME) program. This procedure places emphasis on the FME program and controls. The procedure describes methods for controlling and accounting for material, tools, parts and other foreign material to preclude their uncontrolled introduction in to an open or breached system during work activities. This procedure also provide guidance for establishing and maintaining system cleanliness, recovering from an intrusion of foreign material and re-establishing system cleanliness requirements.

Additionally, procedure, "Containment Inspection (General)," provides detailed guidance for containment inspection to ensure no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of pump suctions during LOCA conditions. This procedure contains an extensive checklist detailing all areas of containment that must be inspected for cleanliness prior to plant startup after each outage.

Procedure, "Containment Inspection (Post Maintenance)," establishes guidance to inventory and control items carried into containment during non-outage entries. This procedure ensures that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of pump suctions during LOCA conditions.

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 2

Vogtle September 2005 Response to Generic Letter 2004-02

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 2

Vogtle September 2005 Response to Generic Letter 2004-02

General System Description:

Vogtle Electric Generating Plant (VEGP) Units 1 and 2 are Westinghouse four loop Pressurized Water Reactor (PWR) design plants. The Residual Heat Removal System (RHR), Chemical Volume and Control System (CVCS – Centrifugal Charging Pump CCP), Safety Injection System (SIS) and Containment Spray System (CSS) pumps are started following a Large Break Loss of Coolant Accident (LBLOCA). Initially, all pumps (2 RHR, 2 CCP, 2 SIS and 2 CSS) take suction from the Refueling Water Storage Tank (RWST). When the RWST level reaches the low-low level set point, the RHR pumps are realigned to take suction from the containment sump. Once the RHR switchover to recirculation is complete, the CCP and SIS pumps take suction from the RHR pump discharge

When the RWST level reaches the empty level alarm, the CSS pumps are realigned to take suction from the containment sump. There are four independent screens (two for RHR and two for CSS) located on elevation 171'-9" in the containment, the lowest floor elevation in the containment exclusive of the reactor cavity, and they are located outside the secondary shield wall.

Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004, requested under "Requested Information," item 2, "Addressees are requested to provide the following information no later than September 1, 2005." The requested information is hereby provided.

GL Item 1:

The 90 day response was provided in SNC letter dated February 25, 2005.

GL Item 2 (a):

Confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.

SNC Response:

Analysis and evaluations are underway to confirm that the Emergency Core Cooling System (ECCS) and CSS recirculation sumps function under debris loading conditions. VEGP will be in compliance by December 31, 2007 with applicable regulatory requirements as identified in Nuclear Energy Institute (NEI) document NEI 04-07, Rev 0, December 2004, "Pressurized Water Reactor Sump Performance Evaluation Methodology" as modified by NRC safety evaluation report issued December 6, 2004.

Major activities planned or completed for SNC include:

Walkdown of containment, for final design change input, is scheduled and will be completed prior to final screen construction.

Debris generation analysis has been completed and the results provided to the transport analysis per the NEI guidance. Containment has mostly fibrous insulation.

Debris transport analysis has been completed showing the debris loading at the sump screens. A Computational Fluid Dynamics (CFD) model was used per the NEI guidance.

NPSH calculations have been reviewed for the RHR and CSS pumps.

Preliminary screen sizing requirements have been determined per the NEI guidance. Verification testing is scheduled.

Downstream effects evaluation is in progress.

SNC is planning to use a passive engineered screen. Screen installations are planned for Fall 2006 for Unit 1 and Spring 2007 for Unit 2.

Modifications are being considered for the safety injection branch line throttle valves, and the RHR pumps backup seals.

GL Item 2 (b):

A general description of and implementation schedule for all corrective actions, including any plant modifications that you identified while responding to this generic letter. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If all corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.

SNC Response:

SNC has outages scheduled for September 17, 2006 on Unit 1 and March 4, 2007 for Unit 2. Required modifications will be completed during these outages, and as such all actions to comply with GL 2004-02 will be completed by December 31, 2007.

Initial analysis activities for debris generation and transport have been completed following the NEI / NRC Guidance. The results of the analysis have been used to develop initial sizing of engineered passive screens for each sump suction. Additional evaluations are under way by Westinghouse to address the possible downstream effects of debris passing through the sump screens. Preliminary chemical effects analysis has been completed. SNC is planning to use the passive engineered screen.

The results of the downstream effects may result in modifications to the safety injection (SI) branch line throttle valves. Modifications may also be required to the RHR pumps backup seals. Any plant modifications required as a result of the downstream effects evaluations will be completed prior to December 31, 2007.

Provided below is the schedule of activities for the completion of analysis, testing and installation of the new engineered screens.

Activity	Common	Unit 1	Unit 2
Walkdown of containment for design considerations		Complete	Fall 2005
Debris generation analysis	Complete		
Debris transport analysis (CFD)	Complete		
Screen sizing per NEI guidance	Complete		
Vendor testing of Screens	Spring 2006		
Downstream effects evaluation including plugging and excessive wear	Spring 2006		
Delivery of engineered screens		Fall 2006	Spring 2007
Installation of new screens and associated plant mods		Fall 2006	Spring 2007
Programs for control of materials in containment are in place. (i.e., insulation, signs and labels, coatings and foreign materials)	In place		

GL Item 2 (c):

A description of the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. The submittal may reference a guidance document (e.g., Regulatory Guide 1.82, Rev. 3, industry guidance) or other methodology previously submitted to the NRC. (The submittal may also reference the response to Item 1 of the Requested Information described above. The documents to be submitted or referenced should include the results of any supporting containment walkdown surveillance performed to identify potential debris sources and other pertinent containment characteristics.)

SNC Response:

SNC has performed analysis to determine the susceptibility of the ECCS and CSS recirculation functions for VEGP to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. These analyses conform to the greatest extent practicable to the NEI 04-07 methodology as approved by the NRC safety evaluation report dated December 6, 2004. Following is a description of the analysis areas performed:

Containment Walkdown

Walkdown of containment was performed by SNC personnel using the guidance of NEI 02-01. The information obtained from the walkdown confirmed the insulation that was installed in containment matched the design documentation. General information was obtained that confirmed the general house keeping condition of containment.

Pipe Break Characterization

The VEGP NSSS system is a four loop Pressurized Water Reactor (PWR). The system consists of one (1) reactor vessel (RX), four (4) steam generators (SGs), four (4) reactor coolant pumps (RCPs), one (1) pressurizer (PZR) and the Reactor Coolant System (RCS) piping. The NSSS system is located inside a bio-shield separated in two (2) halves and one (1) Reactor cavity. Each half houses two (2) steam generators and two (2) RCPs, with loops 1 and 4 occupying the south half and loops 2 and 3 occupying the north half. The outer bio-shield walls extend from the containment base elevation of 171'-9" to the operating floor at El. 220'-0". The walls around the steam generators extend from the operating floor to an elevation of 238'-0".

The piping runs considered for breaks are the RCS hot legs, the RCS cold legs, RCS cross-over legs, and all RCS attached energized piping. Breaks in these lines could decrease RCS inventory and result in the ECCS and/or CSS operating in recirculation mode, in which the system pumps would take suction from the containment sumps. The majority of the insulation inside the VEGP containment is either reflective metal insulation (RMI) or Nukon. The metal reflective insulation is restricted to the reactor, which is located in a closed cavity such that the insulation debris could not be transported by the active pool to the containment sumps. Minor applications of Min-K insulation are installed at locations where clearance is limited, such as wall penetrations. Minor applications of Interam are installed for fire barrier materials. Consequently, few locations would give a different mix of debris types. The largest lines in containment that require evaluation are the hot legs (29-inch ID), the cross-over leg (31-inch ID), the cold leg (27-1/2-inch ID), the pressurizer surge line (16"dia. and 14"dia.), RHR recirculation line to the hot leg (12"dia) and safety injection to the cold leg (10"dia). The SIS hot leg injection, pressurizer spray, charging and letdown lines have a much smaller diameter (6"dia maximum) and are therefore not considered critical.

Since the RHR recirculation lines and the safety injection lines are located within the bio-shield enclosure and are of smaller diameter than the RCS piping, these line breaks would be bounded by the reactor coolant loop breaks and thus do not have to be analyzed. This leaves breaks in the hot legs, the cold legs, cross-over legs and the pressurizer surge line (south half only) for consideration.

Since the north and south halves of the bio-shield enclosure are symmetrical and have the same equipment, and the south half contains more RCS piping, the debris calculations within the bio-shield enclosure is limited to the south half. Results for a particular break location in the south half may be conservatively applied to the north half. The cross-over leg is the largest line (31-inch ID) within the bio-shield enclosure and would produce the largest Zone-of-Influence (ZOI.) Placing a break on the cross-over leg, on the same loop as the pressurizer surge line (loop 4), potentially captures the most insulation debris and is therefore analyzed at two locations. These breaks also impact coatings on sections of the bio-shield wall and floor and potentially capture the most particulate debris.

A hot leg break near the pressurizer surge line is also considered since it can capture the most debris from the two loops (loops 1 and 4). Additionally, a loop 4 cold leg break is analyzed due to its close proximity to the transport path to the recirculation sumps. A hot leg break on loop 1 is also considered near the steam generator nozzle to create a ZOI from the east side of bio-shield enclosure. The pressurizer is located in a cubicle outside the bio-shield enclosure. This cubicle is open on the bottom where the pressurizer is supported on cross beams. The surge line penetrates the west bio-shield wall as 14 inch pipe where it turns upward and connects to the bottom of the pressurizer within a shroud, which limits the amount of debris that a break at the nozzle could generate. A surge line break near the bio-shield wall could impinge on the outer surface of the pressurizer as well as surrounding components in the general area of containment. This is also the closest and most direct break location to the recirculation sumps. A surge line break just outside the bio-shield wall (below the pressurizer cubicle) is therefore also considered.

Therefore, the postulated break locations are as follows:

S1. The Loop 4 Hot Leg near the surge line nozzle at El. 187'-0" [29-inch ID]
S2. The Loop 4 Cross-over Leg at El. 176'-8-1/4" [31-inch ID]
S3. The Loop 1 Hot Leg near the SG nozzle at El. 187'-0" [29-inch ID]
S4. The Loop 4 Cold Leg at El. 187'-0" [27.5-inch ID]
S5. The Loop 3 Cold Leg at El. 187'-0" [27.5-inch ID]
S6. The Surge Line at El. 188'-8" outside the bio-shield [11.188-inch ID]
S7. Alternate Break Methodology (See discussion below)
S8. The Loop 4 Cross-over Leg near the steam generator nozzle [31-inch ID]

Alternate Methodology

For the alternate methodology, the selection of the break size and location in Region I is much simpler. The break size for Region I under the alternate break evaluation is defined as either:

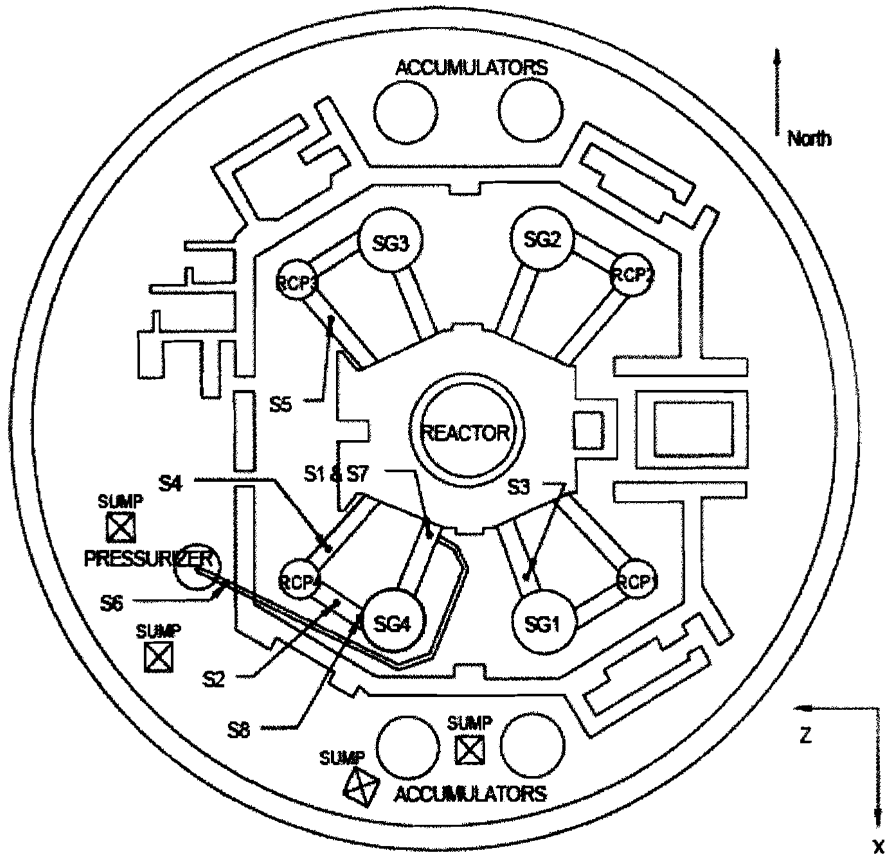
A complete guillotine break of the largest line connected to the RCS piping (16-inch Sch.160 pressurizer surge line 1201-053-16").

OR

A main loop line break equivalent to a guillotine break of a 14-inch Schedule 160 pipe.

As the pressurizer surge line is a 16-inch Sch. 160 (12.812 inch ID) line, this is the size evaluated for the alternate break. For this break (S7), according to the methodology, a double-ended guillotine break is modeled. The location of the break is at the pressurizer surge line connection to the loop 4 hot leg. For Region II of the alternate methodology, the debris quantities are the same as for the deterministic methodology.

Postulated Break Locations



Eight breaks are analyzed within this calculation. They are summarized as follows:

Break Name	Break ID	Elevation	North-South Location	East-West Location	Location
S1	29-inch	187.0 ft	18.0 ft south	7.3 ft west	south bio-shield
S2	31-inch	176.7 ft	29.2 ft south	24.3 ft west	south bio-shield
S3	29-inch	187.0 ft	28.7 ft south	11.6 ft east	south bio-shield
S4	27.5-inch	187.0 ft	17.4 ft south	22.1 ft west	south bio-shield
S5	27.5-inch	187.0 ft	17.4 ft north	22.1 ft west	north bio-shield
S6	11.188-inch	188.7 ft	~ 30 ft south	~40 ft west	below pressurizer cubicle
S7 (alternate)	12.812-inch	187.0 ft	18.0 ft south	7.3 ft west	south bio-shield
S8	31-inch	187.0 ft	32.7 ft south	17.8 ft west	south bio-shield

Notes:
1. North-South and East-West locations are in reference to the Reactor Centerline.
2. Break S5 in the North bio-shield enclosure is modeled at an equivalent location (as Break S4) in the South bio-shield enclosure.

Debris Generation

The debris generation analysis was performed by Sargent and Lundy of Chicago. The analysis determined the debris generated based on the NEI guidance and NRC SER of the NEI guidance. The analysis determined the ZOI for each type of material identified inside containment.

Insulation found inside containment that is adversely affected during a LOCA event, was determined to consist of Nukon fibrous insulation, Min-K, and Interam. Most of the insulation is Nukon. The amount of Min-K and Interam is small.

Qualified coatings are considered in the debris analysis if they are located within a ZOI radius of 10 D. Unqualified coatings are used on various vendor-supplied components throughout containment. The total square footage of unqualified coatings is assumed to be 15,000 ft^2.

A latent debris walkdown/sampling of Unit 1 yielded 60 lbm of debris. This quantity is doubled for analysis purposes.

Latent Debris Accumulation within Containment

Programmatic controls are in place at VEGP that give a solid basis for the amounts of foreign material and latent debris inside containment remaining below the amounts assumed in the sump analysis. These programmatic controls are described in the response to item 2(f).

Debris Transport to the Sump

A debris transport analysis estimated the amount of debris that is transported from debris sources (break locations) to the sump screen.

A debris transport fraction analysis (including erosion) was performed for fibrous debris using a combination of the simple methodology presented in NEI 04-07 and the SER.

All qualified coatings within the ZOI and all unqualified coatings are considered small fines with 100% transport to the sumps.
All other particulate debris is considered small fines with 100% transport to the sumps.

A computational fluid dynamics (CFD) analysis was performed by RWDI Consulting Engineers and Scientists for Sargent and Lundy of Chicago. The CFD modeling techniques used are consistent with the SER, NEI Document number 04-07, and NUREG/CR-6773. CFD analysis of the post-LOCA recirculation containment flows following Break S2 indicate velocities of 0.4 to 1.1 ft/sec can be expected in the containment flow field. These velocities are such that debris within the shield wall would be transported to the openings in the wall. Transport velocities through the wall and to the sumps are of the same order of magnitude.

Head Loss as a Result of Debris Accumulation

The quantity of LOCA generated debris which is transported to the containment sumps for the various postulated breaks was determined based on the methodology of NEI 04-07 and the NRC SER. Once the quantity of debris at the sump screen was known, the head loss across the debris bed on the sump screen was analytically determined. Then the impact of the sump screen debris bed head loss on the NPSH available to the RHR and CSS pumps was evaluated. Head loss parametric testing using plant specific debris mixtures, sump screen configurations, and thermal-hydraulic conditions is planned to finalize the head loss numbers.

Debris Source Term Reduction

There are no plans at this time to reduce the fibrous or particulate debris source term.

Sump Structural Analysis

Structural analysis of the engineered passive screen will be completed when the final screen design is completed. A general description of the screen design is given in the SNC response to item 2 (d) (vii).

Upstream Effects of Debris Accumulation

Evaluations of containment along with review of the CFD model indicate no significant areas will become blocked with debris and hold up water during the sump recirculation phase.

Downstream Effects Associated with any Debris Bypass

The methodologies of NEI 04-07 as modified by the NRC safety evaluation, dated December 6, 2004, and WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," are used to evaluate the downstream effects of debris that is passed by the sump strainer.

GL Item 2 (d):

The submittal should include, at a minimum, the following information:

(i) The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

SNC Response:

Analyses have been completed that show the minimum available NPSH margins with an unblocked sump screen are:

RHR pumps:	15 feet
CSS pumps:	25 feet

(ii) The submerged area of the sump screen at this time and the percent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation.

SNC Response:

All sump screens will be fully submerged (submergence of 100%) before the switchover to recirculation is initiated. The screens will have a nominal submergence of 3 inches at minimum containment water level.

Based on analysis results, the submerged area of the sump is estimated to be 1000 ft^2 for each of the RHR and CSS sump screens. The actual area of the screens will be adjusted based on the results of the vendor screen testing and final head loss margin determination.

(iii) The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS wash down should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates and chemical precipitants caused by chemical reactions in the pool.

SNC Response:

The allowance for head loss from debris accumulation and chemical effects on the submerged sump strainer is 10 ft for RHR and 20 ft for CSS. These numbers allow for 5 foot additional margin for the RHR and CSS sump screens.

Comparisons of the Integrated Chemical Effects Test (ICET) to the VEGP plant specific parameters has been performed and show that the ICET Test # 2 parameters are the closest comparison to VEGP (Test # 2 used trisodium phosphate (TSP) buffer and fiberglass). Based on evaluation of the results to ICET Test # 2, the VEGP design and the VEGP debris loading, the screen design was adjusted to include an increase in the size of the screens to accommodate a 10% increase in head loss across the screens due to chemical effects. Sump screen vendors are currently developing test plans to quantify the additional head loss associated with chemical debris. SNC will incorporate the results of the testing into the final design of the screen as appropriate.

The primary constituents of the debris bed (break S2) are provided below.

Debris Type	Units	Quantity Generated
Fibrous Debris		
NUKON from LOCA	ft^3	1,596.0
NUKON eroded debris	ft^3	129.20
Total Fiber	ft^3	1,725.2
Particulate Debris		
Min-K	ft^3	7.2
Interam	ft^3	19.5
Qualified Coatings	ft^3	30.1
Unqualified Coatings	ft^3	27.8
Total Particulate		84.6
Other Items		
Latent Debris (15% fiber and 85% particulate)	lbm	120
Foreign Material (labels, stickers, etc.)	ft^2	3.6
Chemical Effects	10 % increase in head loss across the screens	

(iv) The basis for concluding that the water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flow paths.

SNC Response:

Evaluations of containment through walk downs and drawing reviews, along with review of the CFD model indicate no significant areas will become blocked with debris and hold up water during the sump recirculation phase. Water sprayed in the refueling canal from the containment sprays will drain back to the elevation of the emergency sump through two 12-in. drain pipes located at the lowest point of the refueling canal. The water passes from the canal to a passageway on the sump floor. The drain piping is isolated during refueling and left open during normal reactor operation. Plant refueling procedures ensure that these drain pipes are opened after refueling prior to plant startup.

(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flow paths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

SNC Response:

The methodologies of NEI 04-07 as modified by the NRC safety evaluation and WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," are being used to evaluate the downstream effects of debris that is passed by the sump strainer. These evaluations are being performed for all components in the recirculation flow paths including, but not limited to, throttle valves, flow orifices, spray nozzles, pumps, heat exchangers, and valves. The following components were identified as requiring additional evaluation:

- ECCS and CSS Pumps (Centrifugal Charging Pumps, SIS Pumps, RHR Pumps, CSS Pumps)
- ECCS and CSS valves including the SI Branch Line Throttle Valves
- ECCS orifices and CSS nozzles
- Piping and instrumentation
- Reactor Vessel internals
- Nuclear Fuel Assemblies

These evaluations are expected to be complete by Spring 2006.

The evaluation of the ECCS orifices, CSS nozzles, piping and instrumentation, and reactor vessel internals indicate that flow restrictions will not occur.

The results of the downstream effects may result in modifications to the safety injection (SI) branch line throttle valves. Modifications may also be required to the RHR pumps backup seals. Any plant modifications required as a result of the downstream effects evaluations will be completed prior to December 31, 2007.

Preliminary evaluation of the nuclear fuel assemblies is still under way.

Adverse gaps or breaches are prohibited by the sump strainer specification, which requires that there shall be no spaces or gaps in the final installation that would allow passage of any particles larger than the screen perforation size.

In addition, technical specifications require that the sump strainers be inspected at least once per 18 months. The inspection procedure currently requires verification that there are no unacceptable holes or gaps in the strainer or between the strainer and adjacent structures and components.

Based on the preliminary reviews performed (with the exception of nuclear fuel), the design of the new screens and the associated design modifications being considered, and the surveillance requirements that are in place, inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flow paths downstream of the sump screen.

(vi) Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.

SNC Response:

See response 2(d)(v).

(vii) Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under predicted flow conditions.

SNC Response:

SNC will install an engineered passive strainer on each RHR and CSS containment sump inlet pipe. The screens will be located outside the secondary shield wall between the shield wall and the containment wall and as such are not exposed to jet impingement or postulated missiles generated from a LOCA event. The screens are of a robust design that will support structural and hydraulic load created by the accumulation of debris during the post LOCA environment. This robust design provides the strength of trash racks and is adequate to protect the screen during a LOCA event.

(viii) If an active approach (e.g., back flushing, powered screens) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

SNC Response:

VEGP is not using the active approach. This section is not applicable to VEGP.

GL Item 2 (e):

A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.

SNC Response:

No additional licensing bases changes have been identified.

GL Item 2 (f):

A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to the extent that their responses address these specific foreign material control issues.

SNC Response:

Insulations

As part of the design change process at SNC, the FSAR, calculations and design specification are reviewed for impacts. Design specifications for thermal insulation for piping and equipment contain the requirements for insulation used inside containment. Design calculations for debris generation will be maintained in the calculations index and thus will be reviewed as design changes are prepared for equipment in containment. The containment sump analysis and a description of the insulation types and amounts assumed in the analysis will be included in the FSAR. Plant procedures control the installation and verification of insulation materials. These procedures provide reference to the guidance provided in the design specifications. Therefore the design change process for SNC will ensure that the assumptions for insulation type and quantity inside containment will be maintained within the analysis assumptions.

Signs and Labels

Procedure, 10016-C, Component Identification, provides the Operations Department instructions for identifying plant components with labels and tags and for placing associated signs within the plant. This procedure will be revised as required to ensure that the appropriate signs and labels are used for identifying components inside containment and that these potential sources of debris introduced into containment will be assessed for potential adverse effects on the ECCS and CSS recirculation functions.

Coatings

Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. Controls were implemented for the procurement, application, and maintenance of Service Level I protective coatings used inside containment in a manner that is consistent with the licensing basis and regulatory requirements applicable to VEGP Units 1 and 2. The requirements of 10 CFR 50, Appendix B, are implemented through specification of appropriate technical and quality requirements for the Service Level I coatings program which includes ongoing maintenance activities. For VEGP, Service Level I coatings are subject to the requirements of NRC Regulatory Guide 1.54; American National Standards Institute (ANSI) N101.2, N101.4, N5.12, and N45.2; VEGP Final Safety Analysis Report (FSAR) section 6.1.2; VEGP Specification X1AJI4.

Coatings used inside the containment, where required, are prequalified coating systems. These coatings are prequalified to the intent of ANSI N101.2 and applicable portions of ANSI N5.12. Quality assurance and documentation requirements of ANSI N45.2-71 and ANSI N101.4 (Class I) are enforced for both coating materials and applications procedures as discussed in FSAR table 6.1.2-1.

VEGP coating applications are governed by plant procedures 25018-C, "Qualification for Painting/Coatings Applications", and 25019-C, "Qualified (N) Coatings". As stated in FSAR section 1.9.54, VEGP conformance to Regulatory Guide 1.54 is discussed in FSAR table 6.1.2-1. Adequate assurance that the applicable requirements for the procurement, application, inspection, and maintenance are implemented is provided by procedures and programmatic controls, approved under the VEGP Quality Assurance Program.

Service Level I coatings used for new applications or repair/replacement activities are procured from a vendor(s) with a quality assurance program meeting the applicable requirements of 10 CFR Part 50, Appendix B. The applicable technical and quality requirements that the vendor is required to meet are specified in procurement documents. Acceptance activities (e.g., receipt inspection, source surveillance, etc.) are conducted in accordance with procedures that are consistent with ANSI N45.2 requirements. This specification of required technical and quality requirements combined with appropriate acceptance activities provides adequate assurance that the coatings received meet the requirements of the procurement documents. The qualification testing of Service Level I coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments referenced above. These coatings, including any substitute coatings, have been evaluated to meet the applicable standards and regulatory requirements previously referenced.

The surface preparation, application and surveillance during installation of Service Level I coatings used for new applications or repair/replacement activities inside containment meets the applicable portions of the standards and regulatory commitments referenced above. Documentation of completion of these activities is performed consistent with the applicable requirements. Where the requirements of the standards and regulatory commitments did not address or were not applicable to repair/replacement activities, these activities were performed in a manner consistent with the generally accepted practices for coatings repair/replacement. These practices are described in various American Society for Testing and Materials (ASTM) standards and coating practice guidelines by industry organizations issued subsequent to those which VEGP has a regulatory commitment. VEGP recognizes that the NRC has not formally endorsed many of the more recent ASTM standards or industry guidelines, but nonetheless, they provide useful information which can be appropriately applied to provide assurance that repair/replacement activities on Service Level 1 coatings are effective in maintaining the acceptability of the coatings. Condition assessments of Service Level I coatings are conducted periodically inside containment at VEGP. Coating condition assessments are conducted as part of an informal, routine coating maintenance program. Presently, coating inspections and evaluations are conducted during outages. The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level I coatings that may be susceptible to detachment from the substrate during a Loss-of-Coolant Accident (LOCA) event is minimized.

Section 6.1.2.1.B.9 of the VEGP FSAR requires that an inventory of unqualified coatings be maintained to ensure appropriate control of coatings inside containment thereby complying with NRC Regulatory Guide (RG) 1.54.

Foreign Materials

Procedure 00254-C, "Foreign Material Exclusion Program", establishes administrative controls and personnel responsibilities for the Foreign Material Exclusion (FME) program. This procedure places emphasis on the FME program, controls, and how they apply to each FME level. This procedure describes methods for controlling and accounting for material, tools, parts, and other foreign material to preclude their uncontrolled introduction into an open or breached system during work activities. This procedure also provide guidance for establishing and maintaining system cleanliness, recovering from an intrusion of foreign material, and re-establishing system cleanliness requirements.

The procedure contains theses specific instructions for work inside containment: "It is critical that materials not be left in containment that will migrate to the Containment Sumps. Temporary materials (mats, cloth, tape, bags, etc.) that will be used inside the Containment Building during refueling outages should be of a color that is easily discernable from walls or other equipment in the Containment Building. When working in Containment, ensure all temporary materials are removed or disposed of properly when an activity is completed."

Procedures 14903-1 and 2, "Containment Emergency Sump Inspection", provide instructions for performing an inspection of the Containment Emergency Sumps prior to returning to power after a refueling. This procedure satisfies the surveillance requirements of Technical Specifications SR 3.5.2.7 and SR 3.5.3.1. This visual inspection verifies each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion. The frequency of this surveillance is at least once every 18 months.

Procedure 14900-C, "Containment Exit Inspection", provides instructions to verify no debris is present in the Containment Building which could be transported to the Containment Sump and cause restriction of Emergency Core Cooling System (ECCS) Pump suctions during LOCA conditions per Technical Requirement Manual TR 13.5.1. This is performed once prior to entry into MODE 4 from MODE 5 and thereafter at the completion of each containment entry.

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 3

Farley List of Regulatory Commitments

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 3

Farley List of Regulatory Commitments

The following table identifies those actions committed by Southern Nuclear Operating Company in this document for Farley Nuclear Plant. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Type		Scheduled Completion Date (If Required)
	One-Time Action	Continuing Compliance	
Installation of Unit 1 and Unit 2 new post LOCA containment sump recirculation screens, completion of required modifications and implementation of required procedural changes.	X		December 31, 2007

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 4

Vogtle List of Regulatory Commitments

Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
September 2005 Response to NRC Generic Letter 2004-02
"Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

Enclosure 4

Vogtle List of Regulatory Commitments

The following table identifies those actions committed by Southern Nuclear Operating Company in this document Vogtle Electric Generating Plant. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Type		Scheduled Completion Date (If Required)
	One-Time Action	Continuing Compliance	
Installation of Unit 1 and Unit 2 new post LOCA containment sump recirculation screens, completion of required modifications and implementation of required procedural changes.	X		December 31, 2007